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W3F1-2004-0102

Timothy G. Mitchell
Director, Engineering
Waterford 3

October 21, 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Supplement to Amendment Request NPF-38-249,
Extended Power Uprate
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES: 1. Entergy Letter dated November 13, 2003, "License Amendment
Request NPF-38-249 Extended Power Uprate"
2. Entergy Letter dated May 7, 2004, "Supplement to Amendment
Request NPF-38-249 Extended Power Uprate"
3. Entergy Letter dated July 14, 2004, "Supplement to Amendment
Request NPF-38-249 Extended Power Uprate"

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Operating License and Technical Specifications to increase the unit's rated thermal power level from 3441 megawatts thermal (MWt) to 3716 MWt.

Entergy, Westinghouse, and members of your staff held a series of calls to discuss various aspects of the Extended Power Uprate (EPU) amendment request previously provided in Reference 1, 2, and 3 including spent fuel pool cooling. As a result of these calls, the responses to six items were determined to need formal response. Entergy's responses to these items are contained in Attachment 1.

There are no technical changes proposed. The original no significant hazards consideration included in Reference 3 is not affected by any information contained in the supplemental letter. There is one new commitment contained in this letter as summarized in Attachment 2.

If you have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on
October 21, 2004.

Sincerely,

A handwritten signature in black ink, appearing to be "TGM/DBM/cbh", written in a cursive style.

TGM/DBM/cbh

Attachments:

1. Additional Information
2. List of Regulatory Commitments

cc: Dr. Bruce S. Mallett
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Attachment 1

To

W3F1-2004-0102

Additional Information

Additional Information

Question 1:

The licensee indicated that the pressurizer relief tank was analyzed and concluded that it will still be capable of performing its function (i.e., condense 791 lbm of steam released from a loss of load event followed by a release of 441 lbm caused by a rod withdrawal incident as the plant returns to power). Based on the results of the licensee's analysis:

- a. How much steam will be released from the loss of load?
- b. How much steam will be released from the rod withdrawal?
- c. What is the impact of the power uprate on the PRT water level/temperature that is required to condense the steam and the pre-EPU vs. post-EPU consequential PRT pressure as compared to the rupture disc design pressure?
- d. What is the impact of the post-EPU steam release on previously analyzed PRT pipe and support temperatures and stress analyses?

Response 1:

Note that while the parenthetical statement is correct regarding the current Final Safety Analysis Report (FSAR) description of the Quench Tank assumed design capability; the Power Uprate Report acknowledged that multiple events are not credible, and thus the events are currently considered individually.

- a. The loss of load transient analysis of record describes a release of less than 1000 lbm.
- b. The transient analysis of record for the CEA withdrawal event at power shows no release, since the safety valve setpoint is not reached.
- c. Because the analyzed events remain within the originally analyzed conditions for the Quench Tank, it is concluded there is no impact as a result of EPU.
- d. Because the analyzed events remain within the originally analyzed conditions for the Quench Tank, it is concluded there is no impact as a result of EPU.

Steam Releases to the Quench Tank

The loss of load event showed an expected increase in steam releases to the quench tank as a result of the increase in core power from the original 3390 MWt to the uprated 3716 MWt.

The Control Element Assembly Withdrawal (CEAW) analysis from high power conditions, which in Cycle 1 predicted a small amount of steam discharge from the pressurizer safety valves, does not result in any discharges for the current analysis performed in support of power uprate. Since the original Cycle 1 analysis, changes have occurred in the reactor protection system which makes the consequences of the high power CEA withdrawal more benign.

At the time of the original construction, the CEA withdrawal from high power was protected by the action of a high power level trip or the Core Protection Calculator (CPC) Departure from Nucleate Boiling (DNBR) trip. The high power level trip had a setpoint in the range of 110% of rated thermal power. The increase in core power resulting from a CEA withdrawal could then potentially go from the initial power level to 110% unless the Core Protection Calculator

System (CPCS), as a result of its DNBR calculation, determined that an earlier reactor trip was needed.

The Cycle 1 analysis of this event went from an initial power of 76% to 109% of rated thermal power before the CPCS DNBR calculation determined that a reactor trip was necessary. Thus a power increase of 33% was possible in the Cycle 1 analysis, a significant addition of excess energy to the Reactor Coolant System (RCS) of steady state initial conditions.

Waterford 3 implemented the COLSSICPCS improvement program beginning with Cycle 2 operation. Among the changes in this program was the addition of a temperature compensated variable overpower trip (VOPT) in the CPCS protection software. This VOPT provided a high power trip ~10% above any steady state power condition. The automatic upward adjustment of this trip was set to a low enough value so that a transient such as a CEA withdrawal would reach the trip setpoint before power increased significantly.

Thus the increase in core power possible from the CEA withdrawal was significantly lower than that which existed at the time of the Cycle 1 analysis. The most adverse case for the high power CEA withdrawal event was no longer from 76% rated thermal power, but from the full power conditions presented in the power uprate report. The increase in core power from initial steady state conditions is now only 10.5%, not the 33% from Cycle 1.

As a result, analysis of the CEA withdrawal, even with the additional rated thermal power, no longer adds enough additional energy over the steady state initial conditions to result in the discharge of steam from the pressurizer safety valves.

Question 2:

Where is the impact of the proposed Extended Power Uprate (EPU) on the condenser hotwell flooding analysis discussed? If it is not discussed, provide a discussion of the evaluation.

Response 2:

FSAR Section 3.6A.6.3 states, "The consequences of flooding in Turbine Building and Fuel Handling Building are not addressed because no equipment essential for safe shutdown is located in these buildings." Therefore, no condenser hotwell flooding analysis is required for EPU.

Question 3:

Describe the impact of the proposed EPU on the water inventory that is required to be maintained in the wet cooling tower basin. Also, explain how this relates to FSAR Table 9.2-10 and include a markup of the table for the staff's review.

Response 3:

As described in PUR Section 2.5.5.3 and RAI Response 6B (W3F1-2004-0035 dated May 7, 2004), the containment heat analysis was re-performed for EPU and the results indicate that the containment heat loads assumed during LOCA conditions are less severe than the current

containment heat load assumed. As a result, the reliance on the WCT to dissipate the accident heat load has been reduced. Therefore, the WCT water inventory required by current Technical Specifications for LOCA is adequate for EPU.

WCT Requirements Post-LOCA Essential Loads (From FSAR 9.2-10)		
	Current	EPU
Evaporation	146,402	86,540
5% Solids	7,320	4,327
Drift	10,667	7,124
Total	164,389	97,991
WCT Capacity (min)	174,000	174,000
Margin	9,611	76,009

Please note that the wet cooling tower basin is the backup water source for the emergency feedwater system.

Question 4:

A markup of the Technical Specification (TS) Bases page for the condensate storage pool indicates that for the situation where the reactor is held in hot shutdown for 4 hours and subsequently cooled down, the water inventory that is needed is reduced by about 100,000 gallons. Explain.

Response 4:

The reduced water consumption is attributable to several factors:

- As a result of a Waterford 3 operating procedure change, the EPU analysis used an upper cooldown rate limit of 50°F per hour in place of the optimum steam generator delta T curve limitations used in the previous analysis.
- Improved modeling of core mixing while in natural circulation. The existing analysis did not account for mixing in the core. As a result, T_{hot} in the hot leg connected to the steam generator having the failed ADV was significantly higher than T_{hot} of the hot leg connected to the steam generator having the functioning ADV. The previous RSB 5-1 analysis was run until the hottest T_{hot} reached 350°F. It has now been demonstrated that during natural circulation there is a great deal of mixing in the core region and the temperature imbalance at the hot legs does not exist. The lower T_{hot} generated by the improved mixing model can be reduced to 350°F sooner.
- Better modeling of the control of plant parameters during the cooldown using the CENTS cooldown controllers.

The RSB 5-1 cooldown was done with CENTS, an NSSS simulation code. The cooldown was conducted using the Waterford 3 procedure for a natural circulation cooldown. The procedure establishes a 50°F per hour upper limit to the cooldown rate. This limits the cooldown rate early in the cooldown. Once the atmospheric dump valve (ADV) has reached

full open, the CENTS code will limit the cooldown rate based on the amount of steam that can be released through the ADV. The flow out the ADV is based on ADV flow area, pressure at the entrance of the ADV and critical flow. As the cooldown progresses, the cooldown rate decreases as steam generator pressure decreases.

Question 5:

The TS Bases for the condensate storage pool establishes a maximum temperature limit of 100 degrees F. Explain why this is not consistent with the maximum temperature limit of 89 degrees F that is imposed on the water stored in the wet cooling tower basin.

Response 5:

The WCT is a component of the Ultimate Heat Sink as described in FSAR Section 9.2.5. Maintaining basin water at or below 89°F ensures the Ultimate Heat Sink can dissipate the maximum post-accident heat load assuming the worst case meteorological conditions. As a backup to the condensate storage pool, the 89°F WCT basin water limit is bounded by the 100°F condensate storage pool TS bases limit since a higher temperature would provide more limiting results for EFW demands.

EPU accident analyses assume 100°F or greater for CSP temperature consistent with the current licensing basis. Note accident analyses are relatively insensitive to this assumption, similar to the discussion provided regarding RWSP temperature in response #2 to W3F1-2004-0037 dated May 12, 2004. CSP temperature is not a setpoint but rather an initial condition. As discussed in TS Bases Insert 3/4.0-1 (Reference Attachment 3 in the July 14, 2004 letter), for less sensitive initial conditions the analytical value is assumed to be the indicated value. This is the case for CSP temperature therefore the value read from the TS indicator for CSP temperature is compared directly to the TS limit of 100°F as indicated in TS Bases Table B 3/4.0-1 (Reference Attachment 3 in the July 14, 2004 letter).

Question 6:

Explain (in a clear and logical manner) specifically what changes to the existing Spent Fuel Pool Cooling licensing basis are being proposed along with the appropriate justification.

Response 6:

As stated in Section 2.5.5.1, "Spent Fuel Pool Cooling and Cleanup System," in Attachment 5 of the November 13, 2003, EPU submittal, the existing spent fuel pool cooling system can handle the decay heat loads associated with the EPU. This conclusion was based on analysis that shows that the post-EPU spent fuel pool temperature will continue to be less than or equal to 155°F following a full core offload and will continue to be less than or equal to 140°F following a partial core offload concurrent with a single failure.

Presented in the accompanying table are the Standard Review Plan (SRP) acceptance criteria and review requirements for the spent fuel pool cooling system. The accompanying table also provides the current Waterford 3 licensing basis responding to the criteria as documented in the Waterford 3 Final Safety Analysis Report (FSAR), the Safety Evaluation

for Amendment 144 (which approved the installation of high density fuel racks), and/or the supporting calculations. The proposed post-EPU licensing basis is also provided for comparison with brief explanations/justifications for differences. The focus is on the cooling aspects since other aspects (e.g., structural integrity, monitoring, testing, etc.) are not impacted by EPU.

Additionally, a table of key inputs and assumptions used in the spent fuel pool cooling analysis is provided. This table compares inputs and assumptions for the current spent fuel pool cooling analysis with the proposed post-EPU spent fuel pool cooling analysis. A justification for differences is provided in the right hand column.

In previous EPU submittals (i.e., November 13, 2003, and May 7, 2004) Entergy indicated that 2116 spent fuel assemblies were assumed to be in the spent fuel storage areas prior to the partial or full core offload. This assumption did not insure that the analysis bounded the total number of fuel assemblies that are allowed to be in the storage areas per Technical Specification 5.6.4. This item has been entered into Entergy's 10CFR50 Appendix B corrective action program at Waterford 3 and the EPU spent fuel cooling analysis has been revised to address this issue. The tables below reflect the revised assumption.

SRP Acceptance Criteria (Subsection II)	SRP Review Requirements (Subsection III)	Current Licensing Basis	Post EPU Licensing Basis
II.1.d. - General Design Criterion 44:	III.1:		
(1) The capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions.	<p>c. The stated quantity of fuel to be cooled by the spent fuel cooling system is consistent with the quantity of fuel stored, as stated in Section 9.1.2 of the SAR.</p> <p>d. For the maximum normal heat load with normal cooling systems in operation, and assuming a single active failure, the temperature of the pool should be kept at or below 140°F and the liquid level in the pool should be maintained.</p>	<p><u>FSAR Section 9.1.2.1(a)</u> The spent fuel storage racks are designed to have storage positions for 2398 fuel assemblies; 1849 in the spent fuel pool; 255 in the cask storage area; and 294 in the refueling canal.</p> <p><u>FSAR Section 9.1.3.1</u> The Fuel Pool System is designed to: a) Remove decay heat produced from a full core placed in the SFP after Rx shutdown, in addition to the decay heat from 2268 previously discharged fuel assemblies (2485 total assemblies).</p> <p>Remove decay heat from 116 assemblies of a core placed in the SFP after reactor shutdown in addition to decay heat from 2369 previously discharged assemblies. With one fuel pump operating, the maximum SFP water temperature will not exceed 140°F.</p>	<p><u>FSAR Section 9.1.2.1(a)</u> No Change.</p> <p><u>FSAR Section 9.1.3.1</u> The Fuel Pool System is designed to: Remove decay heat produced from a full core placed in the SFP after Rx shutdown, in addition to the decay heat from 2224 previously discharged fuel assemblies (2441 total assemblies)</p> <p>This change is acceptable since the amount of discharged assemblies assumed continues to bound the allowable storage capacity given in FSAR Section 9.1.2.1(a) and TS 5.6.4.</p> <p>Remove decay heat from 108 assemblies of a core placed in the SFP after reactor shutdown in addition to decay heat from 2332 previously discharged assemblies (2440 total assemblies). Assuming the most limiting single failure of a divisional electrical bus, the maximum SFP water temperature will not exceed 140°F.</p>

SRP Acceptance Criteria (Subsection II)	SRP Review Requirements (Subsection III)	Current Licensing Basis	Post EPU Licensing Basis
	<p>d. (cont.)For the abnormal maximum heat load (full core unload) the temperature of the pool water should be kept below boiling and the liquid level maintained with normal systems in operation. A single active failure need not be considered for the abnormal case. The associated parameters for the decay heat load of the fuel assemblies, the temperature of the pool water, and the heatup time or rate of pool temperature rise for the stated storage conditions are reviewed on the basis of independent analyses or comparative analyses of pool conditions that have been previously found acceptable.</p>	<p>Remove decay heat from a full core after reactor shutdown in addition to decay heat from 2268 previously discharged assemblies. With two fuel pumps operating, the maximum SFP water temperature will not exceed 155°F.</p>	<p>Remove decay heat from a full core (217 assemblies) after reactor shutdown in addition to decay heat from 2224 previously discharged assemblies (2441 total assemblies). With two fuel pumps operating, the maximum SFP water temperature will not exceed 155°F.</p> <p>The change in the assumed previously discharge assemblies is acceptable since the amount of discharged assemblies assumed continues to bound the allowable storage capacity given in TS 5.6.4.</p> <p>The change to the assumed single failure of a divisional electrical bus is acceptable since it is more limiting in that it takes out a train of redundant cooling components thus reducing the credited heat removal of the system.</p> <p>The current licensing basis assumption of 116 offloaded assemblies was based on a future consideration of 24 month operating cycles. The EPU basis is based on an 18 month operating cycles with a maximum expected offload during a planned outage not to exceed 108 assemblies.</p>

SRP Acceptance Criteria (Subsection II)	SRP Review Requirements (Subsection III)	Current Licensing Basis	Post EPU Licensing Basis
<p>(4) In meeting this criterion, acceptance is based on the recommendations of Branch Technical Position ASB 9-2 for calculating the heat loads and the assumptions set forth in item 1.h of subsection III of this SRP section. The temperature limitations of the pool water identified in item 1.d of subsection III of this SRP section is also used as a basis for meeting this criterion.</p>	<p>h. The calculation for the maximum amount of thermal energy to be removed by the spent fuel cooling system will be made in accordance with Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling" (located in SRP Section 9.2.5) under the following assumed conditions.</p>	<p>Decay heat was determined in accordance with Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling."</p> <p>ASB 9-2 Section B.1 Equation (2) provides the following equation to determine the fraction of operating power:</p> $P/P_o(t_o, t_s) = (1+K) \frac{P/P_o(\infty, t_s) - P/P_o(\infty t_o + t_s)}{P/P_o(\infty t_o + t_s)}$ <p>The uncertainty factor K was only applied to the first term of the equation, therefore the calculated decay heat is conservatively high in the current licensing basis.</p>	<p>Decay heat was determined in accordance with Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling." The background heat loads determined on the current licensing basis were conservatively high based upon an error contained in ASB 9-2. The corrected equation used for EPU is given below.</p> $P/P_o(t_o, t_s) = (1+K) \left[\frac{P/P_o(\infty, t_s) - P/P_o(\infty t_o + t_s)}{P/P_o(\infty t_o + t_s)} \right]$ <p>The fraction of operating power calculated from above benchmarked consistently with the results in Figures 1, 2, and 3 given in ASB 9-2. Additionally, the corrected formula is consistent with ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors." Therefore, it is concluded the uncertainty factor K should be applied to both terms in Equation (2) of ASB 9-2 Section B.1. This formula was previously discussed with the NRC staff.</p>
	<p>i. The uncertainty factor K is set equal to 0.1 for long-term cooling (greater than 107 seconds).</p>	<p>The uncertainty factor K was applied accordingly in accordance with ASB 9-2</p> $K = 0.2 - 0 \text{ sec} < t_s < 1000 \text{ sec}$ $K = 0.1 - t_s \geq 1000 \text{ sec}$	<p>No Change</p>

SRP Acceptance Criteria (Subsection II)	SRP Review Requirements (Subsection III)	Current Licensing Basis	Post EPU Licensing Basis
	<p>ii. The normal maximum spent fuel heat load is set at one refueling load at equilibrium conditions after 150 hours decay and one refueling load to equilibrium conditions after one year decay. (Maximum pool temperature 140°F)</p> <p>iii. The spent fuel pool cooling system should have the capacity to remove the decay heat from one full core at equilibrium conditions after 150 hours decay and one refueling load at equilibrium conditions after 36 days decay, without spent fuel pool bulk water boiling. Cooling system single failure need not be considered concurrent for this condition.</p> <p>iv. For pools with greater than 1-1/3 core capacity, one additional refueling batch at equilibrium conditions after 400 days decay should be included in the cooling requirements.</p>	<p>The normal maximum heat load is set to one refueling load at equilibrium conditions starting after 72 hours plus the background decay heat from 2369 previously discharged assemblies. Offload rate is limited to 4 assemblies per hour.</p> <p>The maximum heat load is set to one refueling load at equilibrium conditions starting after 72 hours plus the background decay heat from 2268 previous discharged assemblies. Offload rate is limited to 4 assemblies per hour.</p> <p>The amount of previous discharged assemblies assumed for the planned and full core offload heat load analyses are:</p> <p>Planned - 2369 assemblies Full Core - 2268 assemblies</p>	<p>The normal maximum heat load is set to one refueling load at equilibrium conditions starting after 72 hours plus the background decay heat from 2332 previously discharged assemblies. Offload rate is limited to 4 assemblies per hour.</p> <p>The total number of assemblies assumed to be discharged for the analysis bounds the number of assemblies (i.e., 2398) allowed per T.S. 5.6.4.</p> <p>The maximum heat load is set to one refueling load at equilibrium conditions starting after 72 hours plus the background decay heat from 2224 previous discharged assemblies. Offload rate is limited to 4 assemblies per hour.</p> <p>The total number of assemblies assumed to be discharged for the analysis bounds the number of assemblies (i.e., 2398) allowed per T.S. 5.6.4.</p> <p>The amount of previous discharged assemblies assumed for the planned and full core offload heat load analyses are:</p> <p>Planned - 2332 assemblies Full Core - 2224 assemblies</p> <p>The total number of assemblies assumed to be discharged for the analysis bounds the number of assemblies (i.e., 2398) allowed per T.S. 5.6.4.</p>

SRP Acceptance Criteria (Subsection II)	SRP Review Requirements (Subsection III)	Current Licensing Basis	Post EPU Licensing Basis
<p>II.1.g - General Design Criterion 61:</p> <p>(4) The capability to prevent reduction in fuel storage coolant inventory under accident conditions in accordance with the guidelines of position C.6 of Regulatory Guide 1.13.</p>		<p><u>Per Amendment #144</u> The NRC evaluated time-to-boil and boil-off rate analysis provided for Waterford 3. This analysis assumed a complete loss of the SFP heat exchangers to cool the SFP with a full core off-load (50.41 MBtu/hr) The minimum time from loss of pool cooling at peak SFP temperature to pool boiling is 2.89 hours with a maximum boil-off rate of 96.48 gpm. This boil-off rate would result in spent fuel uncovered 168 hours after Rx shutdown. The staff found this to be sufficient time for operators to intervene.</p>	<p>No Change: The background decay heat loads determined for the current time to boil analysis were conservatively high based upon an error contained in ASB 9-2 (see discussion above). Correcting the background decay heat loads resulted in a maximum post-EPU heat load that is less than the 50.4 MBtu/hr, therefore the current time to boil analysis bounds EPU.</p>

Inputs/Assumptions for the Spent Fuel Cooling Analysis

Input/Assumption	Current Basis	EPU Basis	Justification
Spent fuel storage limits	<u>TS 5.6.4</u> "The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1849 fuel assemblies in the main pool, 255 fuel assemblies in the cask storage pit and after permanent plant shutdown 294 fuel assemblies in the refueling canal."	<u>TS 5.6.4</u> "The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1849 fuel assemblies in the main pool, 255 fuel assemblies in the cask storage pit and after permanent plant shutdown 294 fuel assemblies in the refueling canal."	No Change
Time core must be subcritical before movement of irradiated fuel can begin.	TS 3.9.3 At least 72 hours	TS 3.9.3 At least 72 hours	No Change
Fuel Pool Temperature Limit	Planned Outage - $\leq 140^{\circ}\text{F}$ Full Core Offload - $\leq 155^{\circ}\text{F}$	Planned Outage - $\leq 140^{\circ}\text{F}$ Full Core Offload - $\leq 155^{\circ}\text{F}$	No Change
Background Decay Heat	<u>Licensed Reactor Power</u> Cycles 1 – 9 – 3390 MWt Cycles 10 up to 2485 Assemblies – 3661.2 MWt	<u>Licensed Reactor Power</u> Cycles 1 – 11 – 3390 MWt Cycles 12 – 13 – 3441 MWt Cycles 14 up to 2441 Assemblies – 3716 MWt	The number of assemblies discharged through Cycle 12 and their associated licensed power level is based on actual data. For Cycle 13 the planned number of assemblies to be discharged during Refuel 13 was assumed at their licensed power level. The maximum number of assemblies expected to be discharged for extended power uprate core designs was assumed thereafter. The total number of assemblies assumed to be discharged for the analysis bounds the number of assemblies (i.e., 2398) allowed per T.S. 5.6.4

Input/Assumption	Current Basis	EPU Basis	Justification
Determination of Decay Heat	<p>Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling"</p> <p>ASB 9-2 Section B.1 Equation (2) provides the following equation to determine the decay fraction of operating power:</p> $P/P_o(t_o, t_s) = (1+K) P/P_o(\infty, t_s) - P/P_o(\infty t_o + t_s)$ <p>The uncertainty factor K was only applied to the first term of the equation, therefore the calculated decay heat is conservatively high in the current licensing basis.</p>	<p>Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling"</p> <p>The corrected equation used for EPU is given below.</p> $P/P_o(t_o, t_s) = (1+K) [P/P_o(\infty, t_s) - P/P_o(\infty t_o + t_s)]$ <p>The fractions of operating power calculated from above were benchmarked consistently with the results to the Figures 1, 2, and 3 given in ASB 9-2. Therefore, it is concluded the uncertainty factor K should be applied to both terms in the Equation (2) of ASB 9-2 Section B.1.</p>	<p>The fractions of operating power calculated from the corrected equation benchmarked consistently with the results in Figures 1, 2, and 3 given in ASB 9-2. Additionally, the corrected formula is consistent with ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors." Therefore, it is concluded the uncertainty factor K should be applied to both terms in Equation (2) of ASB 9-2 Section B.1. This formula was previously discussed with the NRC staff.</p>
Discharged Assemblies – Normal Offload	116 assemblies	108 Assemblies	Current basis assumes a core design for a two year operating cycle. EPU basis assumes maximum amount that will be discharged based on an 18 month operating cycle.
Discharge Assemblies - Rate	4 assemblies/hour	4 assemblies/hour	No Change

Input/Assumption	Current Basis	EPU Basis	Justification
Additional Administrative Controls	None	<p><u>Planned Offload</u> Administrative controls limit the amount of assemblies that can be offloaded as a function of time after shutdown to ensure spent fuel pool temperature is maintained below 140°F assuming a single failure of a divisional electrical bus.</p> <p><u>Full Core Offload</u> Administrative controls limit the amount of assemblies that can be offloaded as a function of time after shutdown to ensure spent fuel pool temperature is maintained below 155°F. A single failure is not considered for the unplanned full core offload.</p>	<p>This is acceptable because these controls maintain spent fuel pool heat loads within the heat removal capability of the spent fuel pool cooling system. Additionally, fuel movement is a highly planned and methodical process well suited for administrative controls.</p> <p>The administrative controls will be maintained in site refueling procedures.</p>
Single Failure	Most efficient spent fuel pool pump	Divisional electrical bus	Divisional electrical bus is more conservative since it takes out a train of redundant cooling components thus reducing the credited heat removal of the system. Procedural controls will be in place to ensure pool heat loads remain within the credited heat removal capacity.
Fuel Pool Heat Exchanger Performance	<p>Planned Outage - 33.7 MBtu/hr</p> <p>Full Core Offload – 50.4 MBtu/hr</p> <p>5% plugged tubes</p>	<p>Planned Outage - 29.1 MBtu/hr</p> <p>Full Core Offload –50.4 MBtu/hr</p> <p>5% plugged tubes</p>	<p>Planned Outage: The reduction in the available heat removal for a planned outage is based on assuming that the single failure is a divisional electrical bus failure. See single failure assumption above.</p> <p>No Change: See full core offload time to boil discussion above.</p> <p>No Change</p>

Input/Assumption	Current Basis	EPU Basis	Justification
CCW Flow for through Fuel Pool Heat Exchanger	Planned Outage - 5000 gpm	Planned Outage - 2768 gpm	The CCW flow to the heat exchanger is reduced as a result of considering a failed CCW Pump during a planned outage (i.e., part of divisional single failure).
	Full Core Offload - 5000 gpm	Full Core Offload - 5000 gpm	No Change
Fuel Pool Cooling Flow through Fuel Pool Heat Exchanger	Planned Outage - 2448 gpm	Planned Outage - 2440 gpm	Flow was conservatively rounded down.
	Full Core Offload - 3650 gpm	Full Core Offload - 3650 gpm	No Change

Attachment 2

To

W3F1-2004-0102

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE- TIME ACTION	CONTINUING COMPLIANCE	
The administrative controls [i.e., to limit the amount of assemblies that can be offloaded as a function of time after shutdown] will be maintained in site refueling procedures.		X	Prior to moving spent fuel from the reactor core following EPU implementation